August 24, 2009

Mr. Christian B. Larsen
Nuclear Vice President & Chief Officer
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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT 1006596, "MATERIALS RELIABILITY PROGRAM (MRP): PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES (MRP-227 – REV. 0) (TAC NO. ME0680)

Dear Mr. Larsen:

By letter dated January 12, 2009, Electric Power Research Institute submitted for U.S. Nuclear Regulatory Commission (NRC) staff review Topical Report (TR) 1006596, "Materials Reliability Program (MRP): Pressurized Water Reactor Internals Inspection and Evaluation Guidelines." Upon review of the information provided, the NRC staff has determined that additional information (RAI) is needed to capture the initial set of technical questions related to the NRC staff's review of TR MRP-227 to support completion of the review. The NRC staff will however issue another set of RAIs based on its review of the MRP-227 report and its supporting reports. By e-mail dated August 22, 2009, Ms. Christine King, Program Manager, MRP, and I agreed that the NRC staff will receive your response to the enclosed RAI questions within 60 days of issuance of this letter. If you have any questions regarding the enclosed RAI questions, please contact me at 301-415-3610.

Sincerely,

**/RA/**

Tanya M. Mensah, Senior Project Manager
Special Projects Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project Nos. 669 and 689

Enclosure:
RAI questions

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION (RAI)

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT 1006596, "MATERIALS RELIABILITY PROGRAM (MRP): PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES

(MRP-227 – REV. 0)

ELECTRIC POWER RESEARCH INSTITUTE

PROJECT NO. 669

In a letter dated January 12, 2009, the Electric Power Research Institute (EPRI) submitted a Topical Report (TR) MRP-227, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines," which addresses the development of an aging management program (AMP) for PWR reactor vessel internal (RVI) components. On July 2, 2009, EPRI provided additional reports that support the technical bases used for developing the AMP, and these reports were submitted to the NRC staff for information only. The NRC staff has reviewed TR MRP-227and developed an initial set of RAIs. Based on further review of the supporting reports, the NRC staff may issue additional RAIs at a later date.

**RAI-1** Many components are placed on a standard 10-year inservice inspection interval coincident with typical American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) inspection requirements. It's not clear, however, whether this 10-year interval is technically acceptable for PWR RVI components. No justification in light of the specific degradation mechanisms being managed has been provided. Other inspection intervals and requirements are based on a certain number of operating cycles. The acceptability of these intervals has also not been established. Please provide a technical justification of the intervals chosen relative to the mechanisms being managed in TR MRP-227.

**RAI-2** In Tables 4-1 through 4-6 and Tables 4-8 and 4-9 of TR MRP-227, the MRP intends to implement visual testing (VT-3) examinations to identify cracking in some PWR RVI components. Historically, enhanced visual testing (EVT-1) or ultrasonic testing (UT) methods are used to effectively identify cracks. Explain why the use of a VT-3 inspection method should be considered acceptable for identifying cracking in some PWR RVI components.

**RAI-3** Eddy current testing (ET) is identified in TR MRP-227 as an inspection method to be used to identify cracking in some PWR RVI components. Clarify whether the acceptance criterion for ET inspections will be based on a "pass – no pass" acceptance criterion (i.e., any ET signals indicating a relevant ET indication would fail the acceptance criterion).

**RAI-4** The accessibility of the primary inspection RVI components is not typically addressed. It is therefore not clear how much inspection coverage is necessary to ensure timely detection of aging effects in the primary inspection RVI components. Discuss whether guidance should be provided in TR MRP-227 regarding minimum inspection volumes/areas which must be achieved to take credit for having effectively inspected a particular RVI component.

ENCLOSURE

**RAI-5** During the extended period of operation, some PWR RVI components are subject to high levels of neutron radiation which may lead to irradiation embrittlement and a loss of fracture toughness and the potential for irradiation-assisted stress corrosion cracking. In combination, these effects may lead to the potential for component failure under some design basis loading conditions. Explain how licensees will be expected to account for potential reduction in fracture toughness when evaluating cracks that are detected during the required inspections, in particular when establishing the frequency of subsequent inspections after cracking is identified.

**RAI-6** Loose parts could be generated due to deterioration of some PWR RVI components during the extended period of operation. Provide information which addresses how the following consequences of loose parts generation were considered in development of the inspection program given in TR MRP-227.

(a) potential for fuel bundle flow blockage and consequential fuel damage,

(b) potential for interference with control rod operation, and

(c) potential for impact damage on reactor internals.

**RAI-7** Alloy 600 PWR RVI components and their associated welds manufactured from Alloys 82 and 182 are susceptible to primary water stress corrosion cracking (PWSCC) when exposed to PWR reactor coolant water. In Table 3-1 of TR MRP-227, the following Babcock and Wilcox (B&W) Alloy X-750 PWR RVI components were welded with Alloy 82 material and yet they were classified under “N” category which excludes inspections for these PWR RVI components:

(1) dowel-to-core barrel cylinder welds, (2) dowel-to-upper grid rib section bottom flange welds, (3) dowel locking welds, (4) dowel-to-guide block welds, and (5) dowel-to-distributor flange welds.

Even though stress levels in these components may not exceed the threshold levels, the NRC staff considers it to be likely that PWSCC can potentially occur due to the introduction of cold work during fabrication. In light of this observation, provide an explanation for excluding inspection requirements for these B&W PWR RVI components.

**RAI-8** When exposed to a light-water reactor temperatures of approximately 500 ºF or higher, the 17-4 precipitation hardened (PH) martensitic stainless steel (MSS) that has previously been subjected to aging (heat treatment) at about 1100 ºF can experience thermal embrittlement and an increase in hardness and a reduction in Charpy V-notch impact test toughness. Operating experience from Oconee Nuclear Station (Information Notice (IN) 2007-02, ADAMS Accession Number ML070100459) shows that thermally embrittled 17-4 PH MSS is susceptible to failure when exposed to unexpected loading conditions. In IN 2007-02, the NRC staff recommended that licensees prevent the deleterious effects of thermal embrittlement in the 17-4 PH MSS components by identifying aging degradation (i.e., cracks), implementing early corrective actions, and monitoring and trending age-related degradation. Therefore, the NRC staff requests that the TR MRP-227 report should include thermal embrittlement as an aging effect for any 17-4 PH MSS RVI components.

**RAI-9** With respect to the management of cast austenitic stainless steel (CASS) aging and embrittlement TR MRP-227 does not appear to address the program's compliance with the requirements specified in the relevant Generic Aging Lessons Learned (GALL) Report AMPs. Provide a discussion of how TR MRP-227 adequately addresses the requirements specified in GALL AMP, XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)," and GALL AMP XI.M13, "Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS)," for CASS materials used in PWR RVI components. Alternatively, if the management CASS PWR RVI component aging is not treated within the scope of TR MRP-227, provide a proposed modification of the report which documents how licensees are expected to manage this mechanism outside of the TR MRP-227 program.

**RAI-10** According to Section A.1.4 in MRP-175, "Materials Reliability Program: PWR Internal Aging Degradation Mechanism Screening Threshold Values," susceptibility to SCC in nickel-based Alloy X-750 PWR RVI components depends on the type of heat treatment that is performed on the alloy. High temperature heat treatment processes that are used on Alloy X-750 components offer better resistance to SCC than the other age hardened heat treatment processes. Licensee determination of the heat treatment applied to their Alloy X-750 PWR RVI components would appear to be a critical parameter in ensuring the licensee's AMP will adequately manage the potential effects of aging. Discuss whether this determination should be included as a license renewal application action item.

**RAI-11** Following on to RAI-10, additional aspects of the TR MRP-227 methodology may need to be addressed by license renewal applicant action items for applications currently under review or those that have yet to be submitted to the NRC. The NRC staff requests the MRP's assistance in identifying potential action items which are: (1) necessary to provide plant-specific information to complete the AMP; (2) necessary to confirm applicant compliance with important assumptions underlying the MRP-227 methodology; or (3) other considerations.

**RAI-12** Provide the loading combinations that were used in determining the peak stress values for any given PWR RVI component. The NRC staff believes that plants that have been implementing power uprates will have to assess whether the peak stress values for any given PWR RVI component are affected by power uprate conditions to determine if their plant is bounded by the assumptions underlying TR MRP-227.

**RAI-13** Certain degradation mechanisms (e.g., void swelling in B&W PWR RVI components) are not inspected for in a particular reactor type. Why does the program not require the most susceptible location for each mechanism in each reactor-type (i.e., B&W, Combustion Engineering, or Westinghouse) be inspected as a primary component to insure that each degradation mechanism is not occurring within the reactor?

**RAI-14** Discuss how the PWR RVI components in each reactor design considered to be the most susceptible to (or most likely to first demonstrate the effects of) a particular degradation mechanism did, or did not, get binned in the primary inspection component group for that design.

**RAI-15** The failure modes, effects, and criticality analysis (FMECA) uses a probabilistic approach with regard to structural stability of any given RVI component and includes the development of a "failure probability factor". What methodology was used to establish the failure probability factor of any RVI component?

**RAI-16** Clarify the conditions under which design basis event (DBE) effects on component performance were considered. How does this approach provide reasonable assurance that the margins against failure are adequately maintained during the license renewal period?

**RAI-17** Component failure due to the same degradation mechanism is not considered to be a common cause failure because of the expectation that damage initiation and growth occurs at different times. However, certain DBEs could potentially lead to a plant condition (damage state) that would not occur unless multiple components were degraded. Discuss how the potential for multi-component failure due to a DBE was considered as part of the development of the MRP-227 program.

**RAI-18** Clarify how plant-specific differences were considered within the FMECA. Discuss whether any additional plant-specific analyses are required, either as a supplement to TR MRP-227 or as indentified plant-specific action items, in order to assure that FMECA analysis supporting the TR MRP-227 program is applicable to a given facility.

**RAI-19** Discuss how a licensee will demonstrate adherence to the reference core loading pattern on a unit-specific basis. Address plant-to-plant variability in neutron flux at various peripheral core locations. Confirm, based on significant operating experience, that "low-leakage" core designs, when normalized by power density, have peripheral neutron fluxes that are consistently within the estimates for the generically studied plants.

**RAI-20** Provide a technical basis to justify the examination acceptance criteria, the sufficiency and relevancy of the links between primary and expansion group components (why were those particular links chosen), and the expansion criteria. Discuss also the technical basis that applied to place certain components in the primary category while others were placed in the expansion category.

**RAI-21** Many of the acceptance criteria provided in TR MRP-227 are vague such as finding “detectable crack-like surface indications,” or “damaged or fractured material,” or “readily detectable cracking.” It’s not clear that these criteria will be uniformly interpreted or implemented from plant to plant. Discuss the need to develop more detailed acceptance criteria on a plant-specific basis and how will the sufficiency of these criteria be established.

**RAI-22** The screening criteria groups materials into susceptibility levels for each degradation mechanism: highly susceptible, moderately susceptible, susceptible, and “below the screening criteria.” Discuss the criteria used to distinguish among the different levels of susceptibility.

**RAI-23** Discuss whether an evaluation was performed for any specific high consequence of failure PWR RVI components such that their inspection might be warranted even in the absence of a currently identifiable mechanism. Are there any PWR RVI components that should be monitored through in-service inspection to protect against unforeseen failure due to the emergence of a potential future degradation mechanism?

**RAI-24** Relevant US and international operating experience with respect to RVI components is not summarized. It is important to indicate what prior RVI component inspections have identified, in particular with respect to justifying the adequacy of existing programs and as part of the basis for the examination requirements (e.g., type, periodicity, importance) identified in MRP-227.

**RAI-25** The cumulative usage factor values for several B&W components need to be confirmed during a comprehensive search of all existing stress and fatigue calculations for the PWR internals. Discuss how such items are intended to translate into plant-specific action items.

**RAI-26** The implications of void swelling are indicated as "dimensional change and distortion…" and it is also noted that "severe void swelling may result in cracking under stress." However, it is not indicated that void swelling can lead to reduced fracture toughness in materials even though it is noted in Section 3.2.7 of TR MRP-227 that "severe swelling (>5%) has been correlated with extremely low fracture toughness values." It is not clear how much void swelling is needed before distortion is detectable via VT-3 examination in susceptible PWR RVI components and whether this threshold for dectectability will also address the concern over potential loss of fracture toughness due to void swelling. Provide a discussion of this topic.